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International Workshop on Plasma-Based Neutron Sources

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US-Russian Federation Workshop on Plasma-Based Device Concepts for
Volumetric Neutron Source for Fusion Nuclear Technology Testing
US-Japan Workshop on Volumetric Neutron Sources
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1.0 Introduction and Workshop Conclusions

The need for the timely development of dedicated neutron sources for nuclear tests of materials, sub-components, and major plasma facing wall and blanket components of fusion reactors has been fully recognized by the community of fusion scientists and engineers, although, so far, financial constraints considerably slowed down the work in this direction. It is probable that, similarly to the history of fission reactors development, a broad range of sources will be needed, ranging from smaller ones suitable for accelerated high-flux, high fluence life-time tests of materials and sub-components to larger devices for relatively low-fluence tests of the major nuclear components of fusion reactors. A high volume, plasma-based neutron source for fusion nuclear testing has been termed a VNS by the IEA. There is no general name for the smaller sources for materials and subcomponent tests; we suggest the name "Accelerated Test Facilities," or ATFs).

Facilities for specimen type material tests (and possibly subcomponent tests) can use accelerator type neutron sources, for example the IFMIF source whose development is now underway within the framework of the IEA. However, the plasma-based sources have the advantages of generating the correct, 14 MeV-based neutron spectrum and providing a larger volume. They would also offer the opportunity of evolutionary development of a broad range of sources, from compact ATFs for material tests to the large-scale VNSes.

The workshop was devoted to discussion of the status and future directions of work on plasma-based neutron sources. The workshop presentations demonstrated significant progress in development of the concepts of these sources and in broadening the required data base.

Two main groups of neutron source designs were presented at the workshop: tokamak-based and mirror-based. Designs of the tokamak-based devices use the extensive data base generated during decades of tokamak research. Their plasma physics performance can be predicted with a high degree of confidence. On the other hand, they are relatively large and expensive, and best suited for VNSes or other large scale test facilities. They also have the advantage of being on the direct path to a power-producing reactor as presently conceived, although alternatives to the tokamak are presently receiving serious consideration for a reactor.

The data base for the mirror-based group of plasma sources is less developed, but they are generally more flexible and, with appropriate selection of parameters, have the potential to be developed as compact ATFs as well as full-scale VNSes.

Also discussed at the workshop were some newly proposed but potentially promising concepts, like those based on the flow-through pinch and electrostatic ion-beam sources.

The studies of the plasma-type neutron sources have been predominantly undertaken in the US, Japan, and Russia (in the latter case, with some involvement of West European partners). The workshop suggests condideration of merging the most promising concepts both within the tokamak group and within the mirror group and of launching joint projects.

In addition to their direct importance for performing numerous specific tests, the neutron sources potentially can have a much broader impact on the fusion program. The design, construction and development of operational procedures for these first fusion-based "instruments" can provide invaluable experience for the work on fusion reactors per se.

There is also a broad range of possible non-fusion applications of the neutron sources: e.g. for burning nuclear wastes, disposal of weapons plutonium, and production of radioactive isotopes. Potentially, these applications may draw additional funding into the development of plasma type neutron sources.

There exists a continuous exchange of information among the groups involved into studies of plasma-type neutron sources. Possible measures directed toward broadening and strengthening this exchange were discussed.

Dr. T. Kawabe suggested that the next workshop be held in Japan, immediately before or after the IAEA Conference that will be held in Yokohama in November, 1998.

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Contents

2.0 Plasma-based neutron source research in the countries participating in the workshop

2.1 US Program

The US fusion program is being restructured in response to Congressional concerns and expected limited resources for the next few years. In the new strategy, fusion energy will be explored within the international context. Materials studies will continue in order to enable attaining the environmental promise of fusion when the program moves to an energy producing reactor. A neutron source will be needed at an appropriate (but undetermined) time to test materials and, eventually, to test reactor modules. It is thus considered important to continue studies into neutron source options for a range of configurations including:

- A VNS high volume, plasma-based neutron source for fusion technology testing complementary to ITER.
- A facility for material irradiation and subcomponent tests, as a possible alternative to accelerator-base neutron sources.
- Neutron applications to other fields than fusion.
- A possible Fusion Engineering Feasibility and Testing Device (FEFT) to address most of the physics and engineering feasibility issues except ignition.

Continued study on various sources to meet these options will place the US in a position to make critical decisions as the strategy of the fusion program evolves.

2.2 Japanese Program

Japan has undertaken activities on Plasma Based Neutron Sources since 1980. The first idea using an open magnetic trap was a cusp-based concept, which was succeeded by mirror-based concept called the Fusion Engineering Facility, FEF. A computer code for designing the neutron source by describing the behavior of the high energy ions in the mirror magnetic field have been developed. After completion of the first study, FEF-II has been pursued, proposed for relatively near-term construction. Recently, the HIEI mirror experimental group joined the neutron source design activity, working with the FEF group. Existing experiments, such as HIEI, as well as GAMMA-10 and international collaboration with GDT, has increased credibility of the performance of those mirror-based neutron sources.

Design studies on the tokamak based neutron source started with an Ultra-low Q tokamak. Recently, a neutron source using the data base of ITER tokamak has been proposed. This activity would both interact with the ITER design work directly and draw on experimental work in existing tokamaks in the world. This idea could be ideal for a VNS for fusion nuclear engineering testing.

An innovative idea of the neutron source based on the inertial, electro-static confinement scheme has been started experimentally, along with a conceptual design study of the neutron source.

From comparative studies between fission reactor development and proposed fusion reactors, it is concluded that several neutron sources will be required. More detailed review studies on the usage of fusion neutrons have been carried out in a committee of the Atomic Energy Society of Japan.

2.3 Russian Federation Program

In Russia, the design and construction of a Gas-Dynamic-Trap neutron source now has a priority just after the ITER program. Russia is open for international collaboration on the development of mirror type neutron source.

The activity of Russia on a neutron source started in 1984, based on the idea of inhomogeneous accumulation of energetic D-T ions in a new type of mirror machine with rather clear physics (so called Gas Dynamic Trap, GDT). Since that time several versions of GDT-based neutron sources were studied (GDT-2, known also as IN-1, and GDT-3). Most recently, an improved version of GDT neutron source (GDT-3-I) was studied. The primary parameters of the latest design of the source are close to those achieved experimentally in earlier mirror machines. This source uses reduced physics and technology parameters (magnetic field strength, injection energy, etc.) making the system more reliable. An initial GDT-based source can have a testing zone area of 1 m^2 with rather small power and tritium consumption (47 MW, 150 g/yr). By adding injection power (with a consistent increase of tritium and power consumption) the area can be increased several times.

The studies of the GDT-based neutron source are being carried out in collaboration with several EC research centers, including FZ Rossendorf (Germany), Laboratory of Ionized Gases (Frascati), Spanish Atomic Energy Commission.

Several tokamak-based VNS projects have been analyzed since 1993 in Russia (Efremov Institute, Kurchatov Institute, and TRINITI). The last of htese projects has been developed in details in Efremov Institute (St. Petersburg). The main features of this design are the physics examined, middle-to-small size (R = 1.7 m), large test surface (more than 10 m^2), normal conducting coils, inner radiation shield availability, solid divertor concept, and a high level of reliability. To reduce power and tritium consumption as well as cost, it is reaonable to decrease the device size (to 1.2 m) when the inner shield is excluded and test surface is reduced (to 5-7 m^2).

3.0 Plasma-based neutron sources – potential uses

Potential uses of plasma-based neutron sources exist in applications to fusion energy for testing materials and for nuclear component and systems testing. In addition, there are several possible applications to non-fusion neutron uses.

3.1 Fusion energy applications

1. Volumetric Neutron Source (VNS). This is a special application specified by the IEA study of an high volume, plasma-based neutron source for fusion nuclear technology testing complementary to ITER.
2. A facility for fusion material irradiation and other aspects of fusion technology development. This facility emphasizes long irradiation time but has a smaller volume requirement than the VNS. In many ways, it is an alternate or extension, e.g. with larger volume, of the accelerator-based neutron sources. We suggest calling such a device an "Accelerated Test Facility," ATF.
3. Fusion Engineering Feasibility and Testing Device (FEFT). This is a device that can potentially address most of the physics and engineering feasibility issues except ignition. It may be suitable for achieving a less costly path for fusion energy development.

3.2 Neutron applications to non-fusion fields

There are a large number of possible applications of an energetic (14 MeV) neutron source. Most of these applications have not been analyzed in detail, but offer opportunities which could be considered as the source technology progresses. They include:

- Fission waste burning (actinides and long-life fission products).
- Weapons plutonium disposition.
- Tritium production.
- Fusion/fission hybrids I: Fissile fuel breeders (fission suppressed) for fission reactors.
- Fusion/fission hybrids II: (a) Driven-fission power reactors
  (b) Driven converters for fission spent fuel.
- Isotope production.

Specific applications of plasma-based neutron sources to non-fusion fields were discussed at the workshop. Analyses of the application of the Gas-Dynamic Trap to a sub-critical reactor and to waste transmutation described one set of options:

Sub-critical fission reactor with the GDT-NS driver. A conceptual scheme of a sub-critical fission reactor with total output power of 500-1000 MW with a GDT-NS as a driver was assessed. The active zone is split into two successive sub-zones separated by a neutron filter. According to preliminary considerations, the power gain factor of this two-cascade blanket was evaluated to be in the range of 250-500.

A GDT-NS used for radioactive waste transmutation. The concept of a low-waste nuclear power facility using a GDT-NS as a transmutator was considered. The facility is comprised of: 3-4 light water reactors each with electric power of 1 GW, a transmutator using a GDT-NS with thermal power of 500 MW, and a local waste repository with moderate capacity. The transmutator is capable of reducing activity of the long - lived actinide wastes by a factor of 10^3.

The Continuous-Flow Z-Pinch was also considered for potential applications:

Applications of the Continuous Flow Z-Pinch (ZPT) to waste burning and tritium production. The workshop heard considerations of preliminary applications of the ZPT to actinide and plutonium burning and to tritium production. These were examples of the applicability of a relatively small fusion neutron to non-fusion neutron systems. At a plasma Q of 0.3, the cost is estimated to be comparable to that of an accelerator-based system; at Q of 1 to 10, the cost is relatively attractive.

A small tokamak VNS could also be used for these applications because of its large test surface (more than 5 m^2) and quite simple procedure of removing and replacing irradiated samples and test units.

4.0 Technical summary and status of proposed devices

Several plasma-based concepts have been considered as possible neutron sources. Detailed summaries of these are included in this section; a short overview is in Table 4.0.

Characteristics of Proposed Plasma-Based Neutron Source Concepts*

	Tokamak (supercon)	Tokamak (normal)	ST	GDT	BPNS	FEF	ZPT
Fusion Power (MW)	300	64	39-59	3	1	10-100	10-500
Neutron wall load (MW/m^2)	1-2	1-2	1	0.5-2	1-5	0.1-5	0.5-50
Total test area (m^2)	200	30	30	0.5-2		5	2-10
Tritium consumption (kg/yr)	17	3.6	2.2-3.3	0.2	0.06	6-60	6-28
Power consumption	200	450	200	50	150	50-80	50-260

*Numbers given here are from workshop presentations and do not necessarily include the full range of possibilities

4.1 Tokamak (superconducting magnets)

A tokamak with superconducting magnets is particularly suitable for a VNS or a FEFT because of its large usable volume of 14 MeV and secondary neutrons. Relative to other concepts, it is large (plasma radius 4 - 5 m) with a rather high tritium consumption (8-10 kg/yr). As a result of requiring a thick (0.8-1.2 m) radiation shield, cryostat, and thermal cryogenic shield, the cost may be as much as 50 % of ITER. The radiation shield thickness may be reduced to 0.5-0.7 m if expensive shield materials (tungsten) are used, but in this case the total cost increases noticeably.

The important advantages of such a neutron source include a relatively low power consumption, primarily supporting the neutral beam injection and cryogenic systems. It would provide operating experience which extrapolates to tokamak reactors. It would extend the physics data base while providing experience in operation of a lithium or molten salt blanket, large superconducting magnets, and the many other technologies (especially nuclear) of a fusion reactor.

A design for a VNS was presented, based on a relatively small extension of the present knowledge of plasma physics and fusion engineering. ITER-relevant design criteria were used for the physics and engineering, resulting in design guidelines:

- A standard tokamak (A > 2) with superconducting toroidal field coils.
- Steady-state operation.
- Neutron wall loading > 1 MW/m^2.
- Tritium breeding with a tritium breeding ratio > 0.8.

Table 4.1.1. Parameters of the VNS based on the superconducting tokamak

	Conventional	Advanced (reversed shear)
Inner shield thickness, m	1.4	1.4
Averaged neutron wall load, MW/m^2	1.0	1.4
Major radius, m	4.5	4.5
Aspect ratio	4.5	4.5
Plasma elongation on 95% flux surface	1.8	1.8
Troyon factor	3.0	3.8
Toroidal field (on axis), T	5.83	5.83
Toroidal field (maximum), T	12.5	12.5
Plasma current, MA	5.6	5.6
Bootstrap current fraction, %	53	72
Neutral beam voltage, MeV	1.0	1.0
Neutral beam power, MW	57.7	28.3
Helium fraction, %	5	5
Total fusion power, MW	300	397
Confinement enhancement factor H (ITER89P)	2.2	2.9
Ion temperature (vol. av.), keV	13.9	13.3
Electron temperature (vol. av.), keV	12.2	12.8
Plasma electron density (vol. av.), 10^{20} m^{-3}	0.96	1.21
Tritium breeding ratio (local, Li-Pb breeder)	1.4	1.4
Site power, MW	400	400
Cost, % of ITER	55	55

Calculations of the tokamak characteristics were considered for a range of parameters, subject to physics constraints such as the ITER transport data base, MHD stability, driven current profiles, and helium ash accumulation. Modeling of the divertor showed reasonable solutions, especially if a radiative divertor can be used. Parameter ranges were explored to determine the optimum operating points for both standard and advanced (reversed shear) designs, including normalized beta (3-4), neutral beam voltage (0.5-1.5 MeV), and confinement H factor (2-3). Blanket concepts were found to yield net tritium breeding ratio of 0.8, with a "makeup" tritium supply of about 2 kg per year required from outside sources to fuel the reacting plasma. Results are summarized in Table 4.1.1.

Issues

- Extrapolation of the physics data base to steady-state operation
- Cost
- Tritium consumption of 2 kG/yr of operation

4.2 Tokamak (normal conducting magnets)

A tokamak neutron source using magnets with normal conductors, can provide a test surface of 5-10 m^2 and neutron load of 1-2 MW/m^2. Reliability of physics, operations and technology should be excellent, with availability more than 40-50 %. Such a device is considerable smaller than a tokamak based on superconducting coils, but still has a large size (plasma radius 1.5-3 m) as well as high level of power (400-800 MW) and tritium consumption (3-6 kg/yr). The possibility of operating in an advanced physics regime offers possible improvements over the conventional scaling.

The tokamak uses multiturn coils (preferable for size, minimizing operational risk and cost), and has a moderate-to-high level of divertor heat loads (near those in ITER) and a moderate-to-thin inner radiation shield. (Device size, fusion and consumption power, and cost decrease significantly with the shield thickness).

Parametric analysis at a neutron flux 1 MW/m^2 shows that with a 0.2 m inboard radiation shield moderate plasma parameters are available (single-null plasma elongation $k \geq 1.7$ and Troyon factor $g_T \geq 2.7$). Higher elongation allows a lower toroidal field, thus reducing technical problems. Parameters of this basic tokamak neutron source option are given in Table 4.2.1.
Plasma modeling yielded the following results:

- Beam energy E_b=140-160 keV (in combination with gas puffing and pellet injection) is reasonable for beam absorption and current drive.
- The VNS operating mode resembles experiments with Supershot and Hot-Ion H-mode: $<T_e>=10$ keV, $<T_i>=30$ keV.
- Ballooning modes are stable; a conducting first wall near the plasma edge ($a_{fw}/a < 1.2$-1.3) and plasma rotation due to neutral beam injection have a profound stabilizing effect on ideal MHD modes stability.

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Table 4.2.1. Main parameters of the basic tokamak, normal conductor option.

Inner shield thickness, m	0.2
Averaged neutron wall load, MW/m^2	1
Major radius, m	1.7
Aspect ratio	3.3
Plasma elongation on 95% flux surface	1.7
Safety factor on 95% flux	3
Troyon factor	2.7
Plasma beta: toroidal, %	3.3
poloidal	0.94
Toroidal field on axis, T	7.6
Plasma current, MA	4.8
Bootstrap current fraction	0.40
Auxiliary heating power, MW	45
Total fusion power, MW	64
Confinement enhancement factor, H	1.85
Plasma temperature, keV	14
Plasma electron density, 10^{20} m^{-3}	1.35
First wall surface, m^2	50
Q factor	1.4
Total flux capacity, V·s	20
Total power supply, MW	650

- Ballooning modes are stable; a conducting first wall near the plasma edge ($a_{fw}/a < 1.2$-1.3) and plasma rotation due to neutral beam injection have a profound stabilizing effect on ideal MHD modes stability.

The impact of energy released during plasma disruptions would be several times less for a VNS than in the ITER EDA. The main reasons are the relatively small values of plasma thermal and magnetic energy (absolute and specific).

A reliable divertor is one of the critical issues in VNS. In the case of the neutron source the divertor problems are rather complicated due to small size of the facility and relatively high power to be exhausted. Reducing the target plate inclination angle to 15° helps to reduce the load to 10 MW/m^2, considered manageable under quasi-steady state operation. A divertor configuration with additional impurity and deuterium gas puff provides reduced target power loads and lowered plasma temperature at their surface.

This VNS may require maintenance associated with the high average neutron load on the entire first wall (more than 1 MW/m^2). It will also require of high tokamak availability (more than 25 %). A normal conducting magnet system with TF coil electric joints should allow a simple assembling/disassembling procedure, making system repair relatively feasible.

Magnet coil design is essentially determined its material behavior under irradiation. This defines an activation of materials, type of insulation, etc. The neutron calculations show, that for fusion neutron flux 1 MW/m^2 on the first wall and 6 years of irradiation the fast neutron fluence ($E_n > 0.1$ MeV) on coils behind the inboard shield is 6×10^{24} m^{-2} and the radiation dose of coil insulation is 5×10^9 Gy. Proposed VNS materials are presented in Table 4.2.2.

Table 4.2.2. Magnet system and shield materials

Shield	Coil insulation	TF coil conductor	PF coil conductor
Tungsten and B_4C	Ceramics	Cr-Zr bronze	Copper

Preliminary analysis of the facility and magnet system design was performed taking into account disassembly, coil materials irradiation, strength and thermal problems, etc. Availability and design reliability requires using vacuum vessel materials with a large data base, such as stainless steel (AISI-316).

The chosen number of TF coils (N=12) allows small toroidal field ripple in plasma region (< 0.5-1%), assuming their outer legs are expanded enough, as well as wide openings between TF coils for blanket modules, NBI channels and divertor blocks. In the PF coils and plasma current scenario the symmetric "demagnetization" of the central solenoid is preferable to smooth out the power peaks if the auxiliary heating system (NBI) is switched only on plasma current plateau. The estimates show that a level of average mechanical stresses in the inner part of TF coils amounts to 200 - 300 MPa with maximum temperature about 80 - 9°C. In the region of TF coil joints the stress and temperature values are less and the joint problem seems to be solvable.

The physical and technical parameters of the design are closely coupled to each other, so that a decrease in total power to 200-400 MW and tritium consumption to 1-2 kg/yr is possible, but with a decrease in device size, test surface, inner radiation shield thickness, and availability (to 20-30%). Such a device would be close to the spherical torus in size (plasma radius = 1-1.5 m) but with conventional physics ($A \geq 2$, $\kappa \leq 2$), and would be based on standard engineering solutions.

Issues

- Extrapolation of physics database to steady-state operation
- Effects of neutron damage
- Power consumption

4.3 Spherical torus (ST)

The spherical torus (ST) is a new approach to confinement based on the very small aspect ratio limit of the tokamak. There are presently small experiments on the ST in England, the US, and Japan, and new, larger experiments are being constructed in England, the US, and Russia to study energy confinement, beta limits, the bootstrap contribution to the plasma current, and other critical physics issues in the device.

The ST is much more compact than the standard tokamak, and thus has potentially lower capital and operating costs. The workshop heard presentations on the latest design of a neutron source based on the concept, using confinement and other physics extrapolated from ITER. The basic parameters of this design are listed in Table 4.3.1, where they are compared with TFTR and JET.

Neutron calculations for this design were also presented, providing confidence in the application to the VNS device. Such parameters as the radial heating profile at the midplane compared well with those expected for the DEMO, although at lower neutron fluxes.

Table 4.3.1. Design parameters for the ST VNS contrasted with TFTR and JET

	VNS (I-II)	TFTR	JET
Major radius, R_0 (m)	0.80	2.9	3.0
Minor radius, a(m)	0.60	0.9	1.2
Aspect ratio, R_0/a	1.33	3.2	2.5
Plasma elongation, κ (=b/a)	2.3	1	1.8
Plasma current, I_p (MA)	10.5-8.6	2.0	5.0
Applied toroidal field at R_0, B_{t0} (T)	1.8	5.0	2.8
Nominal edge safety factor, q_{95}	5.0-5.7	2.5	3.0
Average density, $\langle n_e \rangle$ (10^{20} m^{-3})	1.1-1.8	0.5	0.5
Peak ion temperature, T_{i0} (keV)	24-16	35	30
Heating and current drive pwr (MW)	21-54	35	35
Fusion Power (MW)	39-59	11	20
Duration of D-T burn (s)	s.s.	2	5
Plasma surface area (m^2)	29	100	300
Total energy flux at plasma edge (MW/m^2)	1.0-2.3	0.37	0.13
Heating power to R_0 ratio (MW/m)	36-82	13	13
Average neutron wall load (MW/m^2)	1-1.5	0.1	0.06
Resistive power for TF coil (MW)	100	400	300

Advantages of the ST for neutron production

- Low aspect ratio yields a compact device which potentially may operate at high beta and good energy confinement, with relatively low tritium consumption.
- Data base from the standard tokamak extrapolates favorably to the ST regime.

Issues

- Physics uncertainties (energy confinement, beta, etc.) – These will be studied in the new experiments, with data available within the next few years.
- Center post (toroidal field coil) lifetime in the unshielded neutron flux – The basic issue of radiation embrittlement and increases in resistivity is being determined as part of the ITER project; the present design minimizes stresses on the center post to permit operation with a highly brittle center post.
- High power consumption in center post and in current drive – The extent to which the bootstrap current can alleviate the latter will be determined in upcoming experiments

4.4 Gas dynamic trap (GDT)

Experimental study of plasma stability and confinement in the gas-dynamic trap.

The GDT device has been upgraded since 1993 to achieve significantly improved plasma parameters:

1. Passive stainless steel liner and arc Ti-evaporators were installed inside the central cell enabling the deposition of a Ti film on the liner within 1s before each shot. Additionally, 3 cryopumps were installed. Base pressure was reduced down to 10^{-7} Torr before each plasma shot.

2. Duration of the injection pulse was extended from 0.25 to 1.2 ms.
3. A stabilizing cusp end cell was installed instead of the expander end cell, resulting in more rigid stabilization of the entire plasma.

The main experimental results are the following:

1. Charge-exchange losses of fast ions were significantly reduced. Charge-exchange lifetime of the sloshing ions increased from less then 1 ms to 10 ms. Desorbtion coefficient of the first wall was measured to be 1-1.5.
2. The highest plasma parameters were achieved in the stable regimes of operation with the cusp end cell: electron temperature exceeding 100 eV was observed, sloshing ion density was measured to be higher than 10^{19} m^{-3} near the turning points, target plasma density - $2\text{-}6\times10^{19}$ m^{-3}, plasma beta ~ 10%.
3. Electron temperature and parameters of fast ions are close to those predicted by numerical simulations which account for classical mechanisms only.
4. Longitudinal plasma losses in the regimes with neutral beam heating were found to be in reasonable agreement with theoretical estimates.
5. Thermal insulation of the central cell plasma from the end wall was studied. It was observed that whenever the magnetic field at the end wall becomes smaller than 3-5% of the mirror field the electron heat conduction is suppressed. The measured potentials and electron energies in the expander are consistent with the theory explaining the reduction of the heat conduction by expanding the magnetic field if the field on the end wall is less than $\sqrt{m_e/M_i}$ of the mirror field.
6. No indications of fast ion micro instabilities.

<u>Neutron source simulations in the experiments with synthesized hot ion and electron plasma</u>.

The possibility of simulating the conditions in the neutron source by injecting high power density neutral beams into a target plasma with high electron temperature were considered. The experiments could be done in the GOL-3 facility (initial electron temperature ~400 eV) and in the GDT ($T_e > 100$ eV). The experiments involve construction of an additional local mirror cell with mirror ratio of 1.2-1.3. It is shown that fast ions with density exceeding 10^{20} m^{-3} and mean energy ~25 keV can be provided. In the experiments with synthesized plasma, it is proposed to study non-paraxial and finite beta effects on MHD-stability of the hot-ion "blob" as well as the microstability of the fast ion population. An injector prototype has already been developed and optimal scenarios of synthesized plasma production have been found.

<u>GDT-based neutron source with periodic bursts of neutron flux</u>.

This version of the neutron source may be of interest for studies of relaxation processes in materials as well as for other research projects. To reach this mode of operation, one can add a saw-tooth component to the otherwise steady NBI accelerating voltage. Numerical simulations have shown that, for the 50 microsecond period of the sawtooth, the neutron flux near the turning points can be locally increased by an order of magnitude in periodic bursts with 1-2 microsecond pulse-width.

<u>Energy balance of fast ions in the GDT in the regime with titanium - coated first wall</u>.

The parameters of fast ions and neutral particles in GDT were compared with those predicted by a numerical code. The code simulates fast ion behavior and neutral transport in the GDT by using the Monte-Carlo approach, and predictions agree well with the experimental data. The code was also applied to calculate operational margins of the GDT with increase of the injected neutral beam power. It is shown that fast ion beta as high as 30-70% can be achieved.

Development of a tritium system for the GDT-based neutron source.

Possible options of tritium system of the GDT-based neutron source were considered, and system requirements formulated. General arrangement of the facility was assessed. In order to reduce construction and operational costs, it was proposed to introduce a D-T mixture into the injection system instead of separately introducing D and T. This enables one to avoid isotope separation and reduce the fuel circulation time. The tritium inventory can be thus significantly reduced. Possible design of the three-stage cryopump intended to be used in the tritium system was considered.

Recent findings in the conceptual design of the GDT-based neutron source.

A new version of the GDT-based neutron source, which involves more conservative physics and technical constrains, was considered in the Budker Institute (Novosibirsk). This version uses a fully superconducting magnetic system. To avoid use of warm mirror solenoids, the mirror magnetic field is decreased a factor of 2 to 13T. Electron temperature is also reduced from 1.1 keV to 0.65 keV, only 2.5 times higher than was previously achieved in mirror experiments. It is suggested to reduce the injection voltage to 65 keV to the system more reliable. The simulations show that a neutron flux density of 1.8 MW/m^2 can be achieved in the testing zone of about 1 m^2. The source consumes 47 MW of electric power from the grid and tritium consumption is about 100g yearly. This makes it possible to increase gradually the testing zone area by implementation of more powerful injection system. The reliability of the codes was confirmed by comparison with the experimental results on the GDT device which are found to be in reasonable agreement with the results of computer simulation. The basic parameters of the improved version, GDT-NS (GDT-IMPR) are presented in Table 4.4.1, together with the parameters of the previous version (GDT-3).

Table 4.4.1. GDT neutron source parameters

	GDT-3	GDT-IMPR
Power consumption (MW)	60	47
Neutron flux density (MW/m^2)	2.5	1.8
D^0 beam energy (keV)	80	65
T^0 beam energy	94	65
Electron temperature (keV)	1.1	0.65
"Warm" plasma density (cm^{-3})	2.0×10^{14}	1.7×10^{14}
Mirror-to-mirror size (m)	10	11.4
Injection angle (degrees)	30	30
Mirror field (T)	26	13
Mirror ratio	20	10

A full-scale prototype of the GDT-based NS is now under construction in Budker Institute of Nuclear Physics (Novosibirsk) - so called "Hydrogen Prototype".

Issues.

- The main physics issue of the GDT-based neutron source is the experimental demonstration that the required electron temperature of 500-700 eV can be obtained.
- The main technology issue is that of maintaining a particle balance of a low temperature target plasma, and requires development of high-frequency (a few hundred Hz) pellet injectors.

4.5 Beam-plasma neutron source (BPNS)

Progress has been made in several areas of the BPNS concept, since the last International Neutron Source Workshop and the FEAC Panel 6 Review. This has added to the attractiveness and reliability of the concept. In addition, calculations indicating that an axisymmetric BPNS would be stable to high plasma beta indicates that design of the BPNS and the GDT should be considered together to seek an optimum device. Continued collaboration among researchers interested in the concepts could be highly fruitful.

1. The neutral beam energy was re-optimized from 150 keV to 120 keV by including realistic neutralization efficiencies in addition to ion-confinement and reactivity.
2. Impurity radiation sources were evaluated and determined to have an insignificant effect on BPNS performance. The largest source of impurities was sputtering of sample holders near the hot plasma, but this radiated only 80 kW of the injected 60 MW of neutral beam power, and subjected the sample holders to a surface heating of ≤ 3 MW/m^2 which is within the capabilities of helium gas cooling. Other sources were smaller: limiter sputtering can be neglected for ion energies less than the 120 eV threshold for sputtering tungsten, provided that the heat can be uniformly distributed to <<25-40 MW/m^2 for water cooling(10 MW/m^2 is predicted for axisymmetric limiters), and oxygen impurities are negligible to keep chemical sputtering low. Wall evolved impurities and neutral beam injected impurities both become small in a steady-state device.
3. The plasma column must be terminated (axially) before beta exceeds unity, to prevent the formation of zero magnetic field bubbles within the plasma which would prevent the uniform expansion and power deposition of the plasma.
4. Axisymmetric MHD stability was evaluated and a BPNS hot plasma with a proper magnet design was determined to be flute stable at all beta, and ballooning stable to sufficiently high beta to allow high performance neutron sources as discussed in earlier papers. Axisymmetric operation is very attractive in increasing the volume available for test samples (that was originally occupied partially by a copper minimum-B magnet) and by eliminating the problem of insulator degradation by neutron irradiation in the unshielded copper magnet. The remaining magnets are all superconducting and shielded.
5. Progress in radiative divertors in a number of tokamaks lends credence to the possibility of operation with the plasma column detached from the end wall, and radiation of the neutral beam power by adding impurities near the end walls. Such operation reduces or eliminates end wall sputtering.
6. Evaluation of the PPPL concept, the IDEAL Divertor Simulator, showed that the injection of 10 MW of ICRF power into a "BPNS" scale plasma was feasible. This opens the possibility adding ICRF heating to supplement neutral beam fueling and heating, resulting in lower capital and operating costs, and less charge-exchange induced sputtering.

Issues

- Full physics base not demonstrated and no ongoing experiments.
- Power consumption.

4.6 Fusion Engineering Facility (FEF)

FEF is a tandem mirror device proposed for a neutron and nuclear technology testing facility. Recent design studies suggest a need for a series of such 14 MeV neutron sources based on the development strategy for fusion research and for other applications.

FEF-0 is designed for preliminary irradiation testing for fusion materials as well as fusion engineering research. Material testing, e.g. screening testing of candidate insulators, can be started with this facility at a neutron flux of 0.1-0.5 MW/m^2. Only experimentally proved data bases (most from 2XIIB) are used in this design.

FEF-1, 2 are for fusion material testing, fusion nuclear engineering testing of sub-components, tritium production testing, and thermo-hydraulic testing. They have capability of irradiation of 14 MeV neutrons with a flux of 2 MW/m^2 over an area of 1 m^2. The behavior of sloshing ion has been analyzed by Fokker-Planck simulation. The accumulation of the alpha particles in the mirror magnetic field has been studied by the same code. It was reported that the within the parameter range of this kind of the neutron sources, the accumulated alpha particle would not affect the beta value of the plasma, and thus not affect the MHD stability of the plasma.

A new computer code for the large angle Coulomb scattering is being developing for a non-Maxwellian plasma in the mirror magnetic field with associated with the FEF design.

Detailed design study on the FEF series using the HIEI experimental data is starting. This is characterized by the use of ICRF for heating, production, and stabilization of the plasma in the mirror magnetic field.

Issues

- The mirror data base with full scale operation is incomplete.
- Cost

4.7 Continuous Flow Z-Pinch (ZPT)

The ZPT was presented primarily in the context of its potential application as a neutron source. The concept is based on the interpretation and modeling of old experiments in which a pinch geometry was found to be much more stable to the sausage and kink modes than conventional Z-pinches. Recent MHD computations suggest that this stability may have resulted from velocity shear in the flow of plasma along the pinch. If so, a stable pinch might offer significant opportunities for a neutron source. A linear geometry was described, with the pinch current injected in the center and flowing axially to power exhaust tanks at the ends of the device. This yields a column of neutron producing plasma which can be surrounded by coaxial volumes for neutron source applications such as actinide burning and tritium production.

Advantages of the ZPT

The simple geometry is particularly suited for the utilization of neutrons. If the plasma Q is large enough the resulting cost per neutron is low, resulting in a very attractive system

Issues

- The stabilization physics has not been demonstrated and thus is speculative at this time.
- Other confinement physics, e.g. energy confinement, is not presently fully understood.
- There is no present program (experiments or computations) on the concept, so the physics issues will not be addressed in the near future.

4.8 Alternative approaches

There are a number of alternative (to the tokamak) approaches to fusion which are being explored and which, if developed to the stage that good energy confinement is obtained, are potentially attractive neutron sources. Toroidal configurations include the reversed-field pinch (RFP), the spheromak, and the field-reversed configuration (FRC). Although these were not discussed in detail at the workshop, it will be important to consider them as options as the neutron program develops.

High-power, pulsed sources are also of potential interest as neutron sources, but were not discussed at the workshop.

5. Summary

There has been significant progress in 14 MeV neutron source design since previous meetings.

In conceptual design activities:

- Design of tokamak-based sources has been made more detailed.
- The ST design and neutronics have been extended. It is possible that a tokamak neutron source could operate with sufficiently small tritium consumption that the supply would not be a major problem.
- Design of mirror based sources has been made for the GDT systems; the physics of the BPNS has been extended, indicating that an axisymmetric system is possible.
- FEF design demonstrates a neutron source based on mirror containment for preliminary irradiation testing, based on the existing experimental data base. HIEI mirror experimental data will be used for FEF design, as well as for FEF-II.
- Applications of plasma-based neutron sources for non-fusion purposes have been considered.

In experimental activities:

- Tokamak experiments have demonstrated consistency with ITER scaling laws for confinement.
- Advanced confinement modes have been demonstrated in tokamaks.
- Initial experiments on the ST are promising for good confinement.

- New experimental physics results have been obtained in the Gas Dynamic Trap (GDT) device on MHD stabilization of plasma in axisymmetrical geometry, on sloshing ions accumulation in the trap, and on the suppression of longitudinal electron thermoconductivity.

Important trends include:

- Increasing the 14 MeV neutron flux and testing zone area in mirror (1-4 MW in GDT neutron source case) with the flux density of the level of 1-2 MW/m^2:
- Decreasing the device size and separating the funtions of testing from ignition, resulting in increased technical risk but reducing power and tritium consumption as well as cost in tokamak-based neutron sources.

The mirror-based ATF neutron source will not involve the problems associated with the energetic tail of neutrons with energies more than 14 MeV as detected in the accelerator-based neutron source. Furthermore, due to small power and tritium consumption it can be designed and constructed relatively quickly. Such a neutron source could solve most of the problems of material science and even the testing of some subcomponents of fusion reactor.

A compact tokamak-based neutron source is likely a more expensive installation than an ATF, but it can solve the problems of testing of the fusion reactor components of a large size. We conclude that there is a need for devices of various sizes and costs in order to successfully undertake timely testing of materials, subcomponents, and major nuclear components of a fusion reactor.

We anticipate that various concepts may be merged together to optimize the application to a particular neutron source. An example would be for the proponents of the Gas Dynamic Trap and the Beam Plasma Neutron Source to seek a device which has the best features of each. The possibility of merging and optimizing other device concepts should also be the goal of future research in the field.

In addition to their direct importance for performing numerous specific tests, the neutron sources may have a much broader impact on the fusion program: These sources will be first fusion "instruments" designed and operated as user services, rather than as experiments whose main purpose is to determine operational characteristics; even ITER is conceived as an experimental facility. The very design philosophy of such "instruments" where high reliability and high availability are the dominant features is quite different from that of experimental fusion devices including ITER. The design, construction and development of operational procedures for these first fusion-based "instruments" will provide an invaluable experience for the work on fusion reactors per se.

Formation of Network on Plasma Based Neutron Sources

To increase communication among the researchers who are interested in the plasma based neutron sources and related fields, formation of an network by use of the internet is proposed. The workshop attendees agreed to start this network as a forum for the plasma based neutron sources as soon as possible.

Appendices

A. Workshop participants

Clement Wong
General Atomics

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Alice Ying
UCLA

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TSI Research

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B. Workshop Agenda

Monday, November 18, 1996

8:30	Keith Thomassen	Welcome
8:35	Bick Hooper, LLNL	Organization
	Session Chair: Dmitri Ryutov	
8:45	Sam Berk, OFES (presented by M. Abdou)	Introductory Remarks to the VNS Workshop
	Mohamed Abdou, UCLA	Vision and requirements for VNS
9:30	Edouard Krougliakov, Budker Institute	14 MeV neutron sources for fusion technology studies
10:15	Break	
10:30	Yuichi Ogawa, University of Tokyo	Design of Volumetric Neutron Source based on steady-state tokamak
11:15	Vladimir Filatov, Efremov Institute	Tokamak neutron source design
12:00	Lunch	
1:15	**Group photograph**	
	Session Chair: T. Kawabe	
1:30	Edouard Krougliakov, Budker Institute	Recent findings in the conceptual design of the GDT-based neutron source
2:00	Alexander Ivanov, Budker Institute	Neutron source simulations in the experiments with synthesized hot ion and electron plasma
2:35	Alexander Ivanov, Budker Institute	Experimental study of plasma stability and confinement in the gas-dynamic trap
3:10	Break	
3:25	M. Ohnishi, Kyoto University	Beam-Beam Neutron Source
4:00	Arthur Molvik, LLNL	Beam Plasma Neutron Source revisited
4:40	Edward Cheng, TSI Research, Inc.	A Spherical Torus based Volumetric Neutron Source
5:30	**Initial discussion (Leader: Bick Hooper)**	Applications and opportunities for plasma-based neutron sources
6:30	Workshop dinner	
	Bill Hogan	National Ignition Facility (NIF)

Tuesday, November 19, 1996

	Session Chair: Art Molvik	
9:15	Ralph Cerbone, TSI Research, Inc.	Neutronics Analysis of the ST-VNS
10:00	Alexandr Andriyash, VNIITF	Development of tritium system of the GDT-based neutron source
10:20	Break	
10:35	Alexandr Andriyash, VNIITF	Analysis of possible applications of the GDT-NS for actinides burn-out
		Sub-critical fission reactor with the GDT-NS drive

11:20	Alexander Ivanov, Budker Institute	Energy balance of fast ions in the GDT in the regimes with titanium-coated first wall
12:00	Lunch	
	Session Chair: Alexander Ivanov	
1:00	M. Mizuno, Nihon University	Simulation of Fast Ions in Mirror Based Volumetric Neutron Source
1:35	Y. Yasaka, Kyoto University	Experiment on RF Heating and Control Applicable to Mirror-Based Neutron Source
2:10	Break	
2:20	John Perkins, LLNL	Applications of High Yield Neutron Sources Based on the Flow Through Pinch
2:55	Takaya Kawabe, Tsukuba University	Japanese Activities on Volumetric Neutron Sources, Present and Future
3:45	**Session Chair:** Bick Hooper	Discussion, Workshop Conclusions, and Summary

C. Lessons learned from the Fast Fission Test Facility (FFTF)

(prepared by R. Cerbone)

The Fast Flux Test Facility (FFTF) is a liquid metal (sodium) high energy neutron spectrum when compared with light water reactors. The neutron spectrum is a degraded fission spectrum with a broad maximum extending from 0.10 MeV to 1.0 MeV. The flux at 14 MeV is insignificant for fusion purposes.

FFTF was designed for testing and therefore includes in-core instrumentation capabilities. The testing modules were provided with separate coolant loops and ease of connection. The loops were limited in size, corresponding to a single hexagonal fuel element which measured 12 cm flat to flat. The axial length of the core region was 91.4 cm.

As discussed in Ref. 1, FFTF could have been suitably used for fusion tritium breeding and nuclear heating testing. However because of the inadequate supply of 14 MeV neutrons, FFTF could not provide the proper neutronic environment for the types of testing required for fusion systems. The limited size of the test modules also was a severe handicap.

Thus although FFTF could provide a guide for designing neutron driven test facilities, its primary application was for the Liquid Metal Fast Breeder Reactor (LMFBR); the testing requirements for fusion are so different that the bulk of FFTF experience is of little direct value to fusion.

[1]Modeling, Analysis and Experiments for Fusion Nuclear Technology UCLA-ENG-86-44, FNT-17